ELECTRIC POWER RESEARCH INSTITUTE

MRP Materials Reliability Program MRP 2014-026

(via e-mail)

October 1, 2014

Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Transmittal of EPRI MRP response to the NRC staff related to the *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*

This letter transmits the EPRI MRP response to the following NRC staff question related to MRP-227-A:

"For the fuel alignment plate (FAP) in non-system 80 plants, provide an explanation of how cracking due to SCC and fatigue was determined to be non-significant such that the non-system 80 FAP was assigned to the "No Additional Measures" inspection category." (Transmitted via email from J. Holonich-NRC to K. Amberge-EPRI, 8/14/2014)

The attachment to this letter provides the response to the above question. Should you require any additional information regarding this topic please do not hesitate to contact Kyle Amberge, kamberge@epri.com or 650-804-8037.

Sincerely,

B. C. Rudell
MRP Chairman
Exelon

Anne Demma
MRP Program Manager
EPRI

Cc: Joe Holonich, NRC
James Molkenthin, PWROG Program Manager, Westinghouse

Attachment: Westinghouse Response to the NRC MRP-227 Question on the Combustion Engineering Design Reactor Vessel Internals Fuel Alignment Plate, 9/25/14 (LTR-RIAM-14-85, Rev. 1)

Docket No. 669

D035
NRR

Westinghouse

Attachment to MRP 2014-026
Westinghouse Electric Company
Engineering, Equipment & Major Projects
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To: W. Anthony Nowinowski, Laura L. Genutis

Date: September 25, 2014

cc: Jun C. Bae, John F. Kielb,
Randy G. Lott, Patricia C. Paesano

From: Cheryl L. Boggess
Ext: 412-374-4692

Our ref: LTR-RIAM-14-85, Rev. 1

Subject: **Westinghouse Response to the NRC MRP-227 Question on the Combustion Engineering Design Reactor Vessel Internals Fuel Alignment Plate**

References:
1. Email from Joseph Holonich (U.S. Nuclear Regulatory Commission) to Kyle Amberge (EPRI), "MRP-227 Question for EPRI/Westinghouse on Fuel Alignment Plate," August 4, 2014 (ADAMS ML14217A104).
2. Email from Joseph Holonich (U.S. Nuclear Regulatory Commission) to Kyle Amberge (EPRI), "RE: MRP-227 Question for EPRI/Westinghouse on Fuel Alignment Plate," August 14, 2014 (ADAMS ML14226A945).
3. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863.
4. *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191).* EPRI, Palo Alto, CA: 2006. 1013234.
5. *Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals (MRP-232).* EPRI, Palo Alto, CA: 2008. 1016593.
6. *Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure (MRP-156).* EPRI, Palo Alto, CA: 2005. 1012110.
7. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175).* EPRI, Palo Alto, CA: 2005. 1012081.

Regarding the Combustion Engineering (CE) designed reactor vessel internals fuel alignment plate (FAP), the U.S. Nuclear Regulatory Commission (NRC) has requested a response to the following question (initially posed in [1] and finalized in [2]):

> *"For the fuel alignment plate (FAP) in non-system 80 plants, provide an explanation of how cracking due to SCC and fatigue was determined to be non-significant such that the non-system 80 FAP was assigned to the "No Additional Measures" inspection category."*

The purpose of this letter is to request transmittal of Westinghouse's response to this question to Mr. Kyle Amberge of the Electric Power Research Institute (EPRI). Text has been updated to incorporate customer comments.

An experienced team consisting of utility, original nuclear steam supply system (NSSS) vendors, and independent EPRI experts, representing a broad spectrum of reactor design, operations, and materials expertise, worked on the industry reactor internals aging management project that resulted in an NRC Safety Evaluation and the publication of MRP-227-A [3]. This expert team reviewed available data and industry experience on materials aging to develop a systematic approach to identify and prioritize sampling inspection requirements for reactor vessel internals. As originally summarized in responses to NRC Requests for Additional Information [3] during the MRP-227 regulatory safety review, the process used to develop the MRP-227 recommendations for managing materials aging in reactor vessel internals, including those for the CE FAP, may be described in terms of the following sequence of steps:

Step 1 – Identify Pressurized Water Reactor Internals Components, Materials, and Environments

Step 2 – Identify Degradation Screening Criteria

Step 3 – Characterize Components and Screen for Degradation (A, non-A)

Step 4 – Failure Modes, Effects, and Criticality Analysis (FMECA) Review

Step 5 – Severity categorization (A, B, or C)

Step 6 – Engineering Evaluation and Assessment

Step 7 – Categorize for Inspection (Primary, Expansion, Existing, No Additional Measures) and Aging Management Strategy

Step 8 – Preparation of MRP-227 Inspection and Evaluation Guidelines

The processing of the reactor internals components through these eight steps is detailed in MRP-227-A [3]. The numerical data, screening, and categorization processes for CE internals are detailed in MRP-191 [4] and MRP-232 [5]. Additionally, extensive efforts by the NSSS vendors, key utility personnel, and supporting experts to identify the failure consequences at a component level (as summarized in MRP-156 [6] for Westinghouse and CE) was used extensively in the overall development of susceptibility rankings for the MRP-227 sample inspection guidelines.

Step 2 is of particular note in responding to the current NRC request. Step 2 provided a definition of screening criteria and a screening value basis for the eight degradation mechanisms considered relevant when assessing the impacts of material aging in reactor internals (see Section 1.4 of MRP-175 [7]). The industry defined the screening value to be: "the level of susceptibility when an aging effect may be significant with respect to continued functionality or safety".

Specific screening values were chosen to be sufficiently conservative such that potential component items could be selected for more detailed evaluation of the effects of aging degradation on component functionality.

The eight material degradation criteria to be considered for reactor internals components are:

1. Stress Corrosion Cracking (SCC)
2. Irradiation Assisted Stress Corrosion Cracking
3. Wear
4. Fatigue
5. Thermal Embrittlement

6. Irradiation Embrittlement
7. Void Swelling
8. Irradiation-enhanced Stress Relaxation/Creep

Step 3 in the screening process simply compared the generic component conditions to the MRP-175 screening levels. The CE components with no screened-in aging degradation mechanisms are identified in MRP-191, Table 5-2 [4] and were tentatively placed in Category A, pending review by the FMECA panel in Step 4 of the assessment process. The bounding generic conditions used to assess the CE FAP for all of the FAP design variants in the currently-licensed operating fleet are summarized in MRP-191, Table A-2 [4]. At least one CE FAP design variant originally screened-in during Step 3 of the process for SCC. This was primarily due to the presence of a weld, in addition to wear and fatigue. Subsequent generic ranking of a least one of the CE FAP designs resulted in a "Medium" categorization for both "Likelihood of Failure" and "Likelihood of Damage", yielding placement in FMECA Group 2 at the conclusion of Step 4; see MRP-191, Table 6-6 [4].

A further ranking of the components based on severity of consequence was performed in Step 5 by an expert panel. Each degradation mechanism was separately evaluated by the panel for significance relation to consequences and outcomes. Each was then summed to arrive at a hybrid "score". This ranking process was implemented by the combined consideration of the following factors:

- The extent to which failure might occur due to the degradation mechanism(s) identified; and
- The consequences of such degradation with respect to safety, reliability, and economic risk.

Steps 6 and 7 included detailed evaluations to further define and focus requirements for managing aging. In some cases, comprehensive computer evaluations of components provided detailed data to assist in focusing decisions for future actions. In other instances, investigation of specific variants in the physical design of the individual units resulted in refined screening results. It was determined that a single unique design, the CE System 80, caused the three degradation mechanisms of concern for the FAP to be screened-in. Other CE units did not have the same design characteristics (most notably, structural welds) that would result in exceeding the screening values. A detailed discussion is included in MRP-232, Section 4.1.4.1 [5]. Inspections of the FAP for the non-System 80 units were not required based on the screening results. Therefore, the non-System 80 FAPs were categorized as "No Additional Measures". Degradation mechanisms SCC, wear, and fatigue were subsequently confirmed to be managed by actions on other CE components for the non-System 80 design variants. Management of aging of all eight degradation mechanisms was affirmed.

If there are any questions or clarification needed regarding the aforementioned information, please contact Cheryl Boggess at 412-374-4692.

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